
Experimental Investigation of Convection and Heat Transfer in the Reactor Core for a VHTR

PI: Masahiro Kawaji – City College of New York

Program: NGNP-1

Collaborators: Sanjoy Banerjee – CCNY, Ruel Shinnar – CCNY, Hitesh Bindra – CCNY, Manohar S. Sohal – INL, Richard R. Schultz – INL, Hans Gouger – INL

ABSTRACT

This project aims at identifying and characterizing the conditions under which abnormal heat transfer phenomena would occur in Very High Temperature Reactors (VHTR). High pressure high temperature experiments will be conducted to obtain data that would be used for validation of VHTR design and safety analysis codes. As shown in several Phenomena Identification and Ranking Tables (PIRT) for VHTRs under normal steady-state, transient, and accident scenarios, the key phenomena leading to localized hot spots in the reactor core include degraded heat transfer in coolant channels, laminarization of flow, effects of bypass flow and non-uniform heat generation across the core. These phenomena will be investigated using a unique high pressure, high temperature facility recently constructed at the University. The focus of these experiments will be to generate benchmark data for design and off-design heat transfer for forced, mixed and natural convection in a VHTR core with prismatic blocks.

A key research area related to the VHTR is the development of a best estimate capability to predict coupled convection-radiation heat transfer and calculate the presence of hot spots in the core. This is particularly true given that the bypass flow in a prismatic reactor core may change by as much as a factor of six during the lifetime of the reactor. Hence it is essential to be able to identify and calculate the flow behavior in the core cooling channels and bypass gaps during both operational and accident conditions. To this end, Nuclear Regulatory Commission is modifying or enhancing legacy analysis tools but they either ignore or approximate fundamental physics for both prismatic and pebble bed helium gas cooled reactor cores. Therefore, the main objective of this proposal is to develop reliable thermal-hydraulic data for the development of VHTR design codes and safety analysis codes in coordination with other DOE projects underway or being planned within the Next Generation Nuclear Plant (NGNP) experimental verification & validation (V&V) program.